Perry Buckberg - Entergy LRA letter part 1 of 2

From: "Sanchez, Edward" <esanch1@entergy.com>
To: <phb1@nrc.gov>
Date: 5/1/2007 7:22:38 PM
Subject: Entergy LRA letter part 1 of 2

Perry,

Attached is the first half of the OI response letter. I split it in two due to file size.

Ed Sanchez
Pilgrim Licensing

Mail Envelope Properties (4637CBA4.0E2 : 2 : 61666)

Subject:	Entergy LRA letter part 1 of 2
Creation Date	5/1/2007 7:21:18 PM
From:	"Sanchez, Edward" <esanch1@entergy.com>

Created By:	esanch1@entergy.com

Recipients
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Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

Stephen J. Bethay
Director, Nuclear Assessment

May 1, 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293 License No. DPR-35

License Renewal Application Amendment 16

REFERENCES:
1. Entergy Letter, License Renewal Application, dated January 25, 2006
2. NRC Safety Evaluation Report with Open Items Related to the Pilgrim License Renewal Application, dated March 2007
3. NRC Request for additional information for review of the Pilgrim license renewal application, dated March 26, 2007
4. Entergy Letter, Comments on NRC Draft Safety Evaluation Report Related to PNPS LRA, dated March 28, 2007

LETTER NUMBER: 2.07.027

Dear Sir or Madam:

In Reference 1, Entergy Nuclear Operations, Inc. applied for renewal of the Pilgrim Nuclear Power Station operating license. NRC TAC No. MC9669 was assigned to the application.

This letter provides information to address the Open Items from the NRC safety evaluation report (SER), (Reference 2). This letter also provides information in response to a request for additional information (Reference 3) related to Open Item 4.2. In addition, this letter includes LRA amendments resulting from review of the NRC SER.

Commitments made by this letter are contained in Attachment A.

Please contact Mr. Bryan Ford, (508) 830-8403, if you have questions regarding this subject.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 1, 2007.

Sincerely,

Stephen J. Bethay

Stephen J. Bethay
Director Nuclear Safety Assessment

ERS/dl

Attachments:

Attachment A:	Revised List of Regulatory Commitments
Attachment B:	Information in Response to the Open Items Listed in the Draft NRC SER, Including Associated LRA Amendments and License Condition
Attachment C:	LRA Amendments to Delete the BWRVIP-48 and BWRVIP-49 Fatigue Assessments as TLAAs
Attachment D:	Torus Room Concrete Base Mat Evaluation (Dr. Franz Ulm, M.I.T.)
Attachment E:	Structural Integrity Associates Fluence Evaluation for PNPS

cc: see next page

cc: with Attachments

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ATTACHMENT A to Letter 2.07.027
(8 pages)

Revised List of Regulatory Commitments

Revised List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
1	Implement the Buried Piping and Tanks Inspection Program as described in LRA Section B.1.2.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.2 / Audit Item 320
2	Enhance the implementing procedure for ASME Section XI inservice inspection and testing to specify that the guidelines in Generic Letter 88-01 or approved BWRVIP-75 shall be considered in determining sample expansion if indications are found in Generic Letter 88-01 welds.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.6 / Audit Item 320
3	Inspect fifteen (15) percent of the top guide locations using enhanced visual inspection technique, EVT-1, within the first 18 years of the period of extended operation, with at least one-third of the inspections to be completed within the first six (6) years and at least two-thirds within the first 12 years of the period of extended operations. Locations selected for examination will be areas that have exceeded the neutron fluence threshold.	As stated in the commitment.	Letters 2.06.003 and 2.06.057 and 2.06.064 and 2.06.081	B.1.8 / Audit Items 155, 320
4	Enhance the Diesel Fuel Monitoring Program to include quarterly sampling of the security diesel generator fuel storage tank. Particulates (filterable solids), water and sediment checks will be performed on the samples. Filterable solids acceptance criteria will be = 10 mg/l. Water and sediment acceptance criteria will be = 0.05%.	June 8, 2012	Letters 2.06.003 and 2.06.057 and 2.06.089	B.1.10 / Audit Items 320, 566
5	Enhance the Diesel Fuel Monitoring Program to install instrumentation to monitor for leakage between the two walls of the security diesel generator fuel storage tank to ensure that significant degradation is not occurring.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.10 / Audit Items 155, 320
6	Enhance the Diesel Fuel Monitoring Program to specify acceptance criterion for UT measurements of emergency diesel generator fuel storage tanks (T-126A&B).	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.10 / Audit Items 165, 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
7	Enhance Fire Protection Program procedures to state that the diesel engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be enhanced to verify that the diesel engine did not exhibit signs of degradation while it was running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Also, enhance procedures to clarify that the diesel-driven fire pump engine is inspected for evidence of corrosion in the intake air, turbocharger, and jacket water system components as well as lube oil cooler. The jacket water heat exchanger is inspected for evidence of corrosion or buildup to manage loss of material and fouling on the tubes. Also, the engine exhaust piping and silencer are inspected for evidence of internal corrosion or cracking.	June 8, 2012	Letters 2.06.003 and 2.06.057 and 2.06.064	B.1.13.1 / Audit Items 320, 378
8	Enhance the Fire Protection Program procedure for Halon system functional testing to state that the Halon 1301 flex hoses shall be replaced if leakage occurs during the system functional test.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.13.1 / Audit Item 320
9	Enhance Fire Water System Program procedures to include inspection of hose reels for corrosion. Acceptance criteria will be enhanced to verify no significant corrosion.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.13.2 / Audit Item 320
10	Enhance the Fire Water System Program to state that a sample of sprinkler heads will be inspected using guidance of NFPA 25 (2002 Edition) Section 5.3.1.1.1. NFPA 25 also contains guidance to repeat this sampling every 10 years after initial field service testing.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.13.2 / Audit Item 320
11	Enhance the Fire Water System Program to state that wall thickness evaluations of fire protection piping will be performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.13.2 / Audit Item 320
12	Implement the Heat Exchanger Monitoring Program as described in LRA Section B.1.15.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.15 / Audit Item 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
13	Enhance the Instrument Air Quality Program to include a sample point in the standby gas treatment and torus vacuum breaker instrument air subsystem in addition to the instrument air header sample points.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.17 / Audit Item 320
14	Implement the Metal-Enclosed Bus Inspection Program as described in LRA Section B.1.18.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.18 / Audit Item 320
15	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program as described in LRA Section B.1.19. Include developing a formal procedure to inspect manholes for in-scope medium voltage cable.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.19 / Audit items 311, 320
16	Implement the Non-EQ Instrumentation Circuits Test Review Program as described in LRA Section B.1.20.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.20 / Audit Item 320
17	Implement the Non-EQ Insulated Cables and Connections Program as described in LRA Section B.1.21.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.21 / Audit Item 320
18	Enhance the Oil Analysis Program to periodically change CRD pump lubricating oil. A particle count and check for water will be performed on the drained oil to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.22 / Audit Item 320
19	Enhance Oil Analysis Program procedures for security diesel and reactor water cleanup pump oil changes to obtain oil samples from the drained oil. Procedures for lubricating oil analysis will be enhanced to specify that a particle count and check for water are performed on oil samples from the fire water pump diesel, security diesel, and reactor water cleanup pumps.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.22 / Audit Item 320
20	Implement the One-Time Inspection Program as described in LRA Section B.1.23.	June 8, 2012	Letters 2.06.003 and 2.06.057 and 2.07.023	B.1.23 / Audit Items 219, 320
21	Enhance the Periodic Surveillance and Preventive Maintenance Program as necessary to assure that the effects of aging will be managed as described in LRA Section B.1.24.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.24 / Audit Item 320

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
22	Enhance the Reactor Vessel Surveillance Program to proceduralize the data analysis, acceptance criteria, and corrective actions described in LRA Section B.1.26.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.26 / Audit Item 320
23	Implement the Selective Leaching Program in accordance with the program as described in LRA Section B.1.27.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.27 / Audit Item 320
24	Enhance the Service Water Integrity Program procedure to clarify that heat transfer test results are trended.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.28 / Audit Item 320
25	Enhance the Structures Monitoring Program procedure to clarify that the discharge structure, security diesel generator building, trenches, valve pits, manholes, duct banks, underground fuel oil tank foundations, manway seals and gaskets, hatch seals and gaskets, underwater concrete in the intake structure, and crane rails and girders are included in the program. In addition, the Structures Monitoring Program will be revised to require opportunistic inspections of inaccessible concrete areas when they become accessible.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.29.2 / Audit Items 238, 320
26	Enhance Structures Monitoring Program guidance for performing structural examinations of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.29.2 / Audit Item 320
27	Enhance the Water Control Structures Monitoring Program scope to include the east breakwater, jetties, and onshore revetments in addition to the main breakwater.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.29.3 / Audit Item 320
28	Enhance System Walkdown Program guidance documents to perform periodic system engineer inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.30 / Audit Items 320, 327

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
29	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.1.31.	June 8, 2012	Letters 2.06.003 and 2.06.057	B.1.31 / Audit Items 257, 320
30	Perform a code repair of the CRD return line nozzle to cap weld if the installed weld repair is not approved via accepted code cases, revised codes, or an approved relief request for subsequent inspection intervals.	June 30, 2015	Letter 2.06.057	B.1.3 / Audit Items 141, 320
31	At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the PNPS vintage, PNPS will implement one or more of the following: (1) Refine the fatigue analyses to determine valid CUFs less than 1 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF. 2. More limiting PNPS-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations. 3. Representative CUF values from other plants, adjusted to or enveloping the PNPS plant specific external loads may be used if demonstrated applicable to PNPS. 4. An analysis using an NRC-approved version of the ASME code of NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. The determination of Fen will account for operating times with both hydrogen water chemistry and normal water chemistry. (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC). (3) Repair or replace the affected locations before exceeding a CUF of 1.0. Should PNPS select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.	June 8, 2012 June 8, 2010 for submitting the aging management program if PNPS selects the option of managing the affects of aging due to environmentally assisted fatigue.	Letters 2.06.057 and 2.06.064 and 2.06.081 and 2.07.005	4.3.3 / Audit Items 302, 346

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
32	Implement the enhanced Bolting Integrity Program described in Attachment C of Pilgrim License Renewal Application Amendment 5 (Letter 2.06.064).	June 8, 2012	Letters 2.06.057 and 2.06.064 and 2.06.081	Audit items 364, 373, 389, 390, 432, 443, 470
33	PNPS will inspect the inaccessible jet pump thermal sleeve and core spray thermal sleeve welds if and when the necessary technique and equipment become available and the technique is demonstrated by the vendor, including delivery system.	As stated in the commitment.	Letter 2.06.057	Audit Items 320, 488
34	Within the first 6 years of the period of extended operation and every 12 years thereafter, PNPS will inspect the access hole covers with UT methods. Alternatively, PNPS will inspect the access hole covers in accordance with BWRVIP guidelines should such guidance become available.	June 8, 2018	Letters 2.06.057 and 2.06.089	Audit Items 320, 461
35	At least 2 years prior to entering the period of extended operation, for reactor vessel components, including the feedwater nozzles, PNPS will implement one or more of the following: (1) Refine the fatigue analyses to determine valid CUFs less than 1. Determine valid CUFs based on numbers of transient cycles projected to be valid for the period of extended operation. Determine CUFs in accordance with an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC). (3) Repair of replace the affected locations before exceeding a CUF of 1.0. Should PNPS select the option to manage the aging effects due to fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.	June 8, 2012 June 8, 2010 for submitting the aging management program if PNPS selects the option of managing the affects of aging.	Letters 2.06.057 and 2.06.064 and 2.06.081	Audit Item 345

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
36	To ensure that significant degradation on the bottom of the condensate storage tank is not occurring, a one-time ultrasonic thickness examination in accessible areas of the bottom of the condensate storage tank will be performed. Standard examination and sampling techniques will be utilized.	June 8, 2012	Letter 2.06.057	Audit Items 320, 363
37	The BWR Vessel Internals Program includes inspections of the steam dryer. Inspections of the steam dryer will follow the guidelines of BWRVIP-139 and General Electric SIL 644 Rev. 1.	June 8, 2012	Letter 2.06.089	A.2.1.8 / Conference call on September 25, 2006
38	Enhance the Diesel Fuel Monitoring Program to include periodic ultrasonic thickness measurement of the bottom surface of the diesel fire pump day tank. The first ultrasonic inspection of the bottom surface of the diesel fire pump day tank will occur prior to the period of extended operation, following engineering analysis to determine acceptance criteria and test locations. Subsequent test intervals will be determined based on the first inspection results.	June 8, 2012	Letter 2.06.089	B.1.10 / Audit Item 565
39	Perform a one-time inspection of the Main Stack foundation prior to the period of extended operation.	June 8, 2012	Letter 2.06.094	B.1.23 / Audit Item 581
40	Enhance the Oil Analysis Program by documenting program elements 1 through 7 in controlled documents. The program elements will include enhancements identified in the PNPS license renewal application and subsequent amendments to the application. The program will include periodic sampling for the parameters specified under the Parameters Monitored/Inspected attribute of NUREG-1801 Section XI.M39, Lubricating Oil Analysis. The controlled documents will specify appropriate acceptance criteria and corrective actions in the event acceptance criteria are not met. The basis for acceptance criteria will be defined.	June 8, 2012	Letter 2.06.094	B.1.22 / Audit Items 553 and 589
41	Enhance the Containment Inservice Inspection (CII) Program to require augmented inspection in accordance with ASME Section XI IWE-1240, of the drywell shell adjacent to the sand cushion following indications of water leakage into the annulus air gap.	June 8, 2012	Letter 2.06.094	A.2.1.17 and B.1.16.1
42	Implement the Bolted Cable Connections Program, described in Attachment C of Pilgrim License Renewal Application 11 (Letter 2.07.003), prior to the period of extended operation.	June 8, 2012	Letter 2.07.003	A.2.1.40 and B.1.34

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	Related LRA Section No./ Comments
43	Include within the Structures Monitoring Program provisions to ensure groundwater samples are evaluated periodically to assess the aggressiveness of groundwater to concrete, as described in Attachment E of LRA Amendment 12 (Letter 2.07.005), prior to the period of extended operation.	June 8, 2012	Letter 2.07.005	A.2.1.32 and B.1.29.2
44	Perform another set of the UT measurements just above and adjacent to the sand cushion region prior to the period of extended operation and once within the first 10 years of the period of extended operation.	As stated in the commitment.	Letter 2.07.010	A.2.1.17 and B.1.16.1
45	If groundwater continues to collect on the Torus Room floor, obtain samples and test such water to determine its pH and verify the water is non-aggressive as defined in NUREG-1801 Section III.A1 item III.A.1-4 once prior to the period of extended operation and once within the first ten years of the period of extended operation.	As stated in the commitment.	Letters 2.07.010 and 2.07.027	A.2.1.32 and B.1.29.2
46	Inspect the condition of a sample of the torus hold-down bolts and associated grout and determine appropriate actions based on the findings prior to the period of extended operation.	June 8, 2012	Letter 2.07.027	A.2.1.32 and B.1.29.2
47	Submit to the NRC an action plan to improve benchmarking data to support approval of new P-T curves for Pilgrim.	Sept.15, 2007	Letter 2.07.027	4.2.2, A.2.2.1.1, and A.2.2.1.2
48	On or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.	June 8, 2010	Letter 2.07.027	4.2, 4.7.1, A.1.1 and A.2.2.1

ATTACHMENT B to Letter 2.07.027

(17 pages)

Information in Response to the Open Items Listed in the Draft NRC SER, Including Associated LRA Amendments and License Condition

OI 2.3.3.6: (SER Section 2.3.3.6 - Security Diesel)

LRA Table 2.3.3-6 shows the component types subject to an AMR but the security diesel system was not in the FSAR or in any license renewal drawings; therefore, the staff could not determine the portion of the security diesel system within the scope of license renewal. Additionally, the staff could not determine whether any components within the scope of license renewal were not shown as subject to an AMR. The staff referred this issue to NRC Region I who will determine whether security diesel system components are within the scope of license renewal.

OI 2.3.3.6 Response

Entergy provided NRC Region I with support as requested.

OI 3.0.3.2.10: (SER Section 3.0.3.2.10 - Fire Protection Program)

The applicant is taking an exception to the GALL Report program element "detection of aging effects," specifically:

The NUREG-1801 program states that approximately 10 percent of each type of penetration seal should be visually inspected at least once every refueling outage. The PNPS program specifies inspection of approximately 20 percent of the seals each operating cycle, with all accessible fire barrier penetration seals being inspected at least once every five operating cycles.

The LRA states that, because aging effects typically are manifested over several years, this variation in inspection frequency is insignificant. GALL AMP XI.M26 specifies approximately 10 percent of each type of seal should be inspected visually at least every refueling outage (two years). The applicant clarified that the program specifies inspection of approximately 20 percent of the seals, including at least one seal of each type, each operating cycle, with all accessible fire barrier penetration seals being inspected at least once every five operating cycles. The applicant needs to address how to manage the aging effect of inaccessible fire barrier penetration seals.

OI 3.0.3.2.10 Response

The PNPS requirement to inspect penetration seals applies to 100% of the seals. The word "accessible" is not necessary in the discussion of the exception for Detection of Aging Effects in the PNPS program. All fire barrier penetration seals are inspected at least once every five operating cycles. In LRA Appendix B, Section B.1.13.1, the word "accessible" is removed resulting in the following description of the exception for Detection of Aging Effects.

> The NUREG-1801 program states that approximately 10% of each type of penetration seal should be visually inspected at least once every refueling outage. The PNPS program specifies inspection of approximately 20% of the seals each operating cycle, with all ~~accessible~~ fire barrier penetration seals being inspected at least once every five operating cycles.

OI 3.0.3.3.2: (SER Section 3.0.3.3.2 - Containment Inservice Inspection and Section 3.5.2.2.1 - PWR and BWR Containments)

A recent NRC Region 1 inspection team observations indicated the following:

- The flow switch in the bellows rupture drain had failed its surveillance in December 2005 and has not been fixed or evaluated. In addition, the flow switch also failed in 1999.
- Monitoring of other drains has been inconclusive and not well documented.
- The torus room floor has had water on the floor on multiple occasions.

In Request for Additional Information (RAI) B.1.16.1, dated November 7, 2006, the applicant was asked to address the above finding and discuss the impact on the aging management of potential loss of material due to corrosion in the inaccessible area of the Mark I steel containment drywell shell, basemat, including the sand pocket region for the period of extended operation.

OI 3.0.3.3.2 Response

Entergy letter dated March 13, 2007 provided information to address this open item and RAI B.1.16.1. With regard to the issue of water on the torus room floor, Attachment D to this letter contains a report prepared by a consultant to Entergy that provides a detailed evaluation of the groundwater seepage through the concrete basemat.

Commitments 43, 45, and 46 will be implemented to address this issue. Commitment 45 made in the March 13, 2007 Entergy letter is revised by this letter to require it be performed once within the first ten years of the period of extended operation in addition to it being performed once prior to the period of extended operation. Commitment 46 is added by this letter. These commitments are listed in Attachment A to this letter and read as follows:

43	Include within the Structures Monitoring Program provisions to ensure groundwater samples are evaluated periodically to assess the aggressiveness of groundwater to concrete, as described in Attachment E of LRA Amendment 12 (Letter 2.07.005), prior to the period of extended operation.
45	If groundwater continues to collect on the Torus Room floor, obtain samples and test such water to determine its pH and verify the water is non-aggressive as defined in NUREG-1801 Section III.A1 item III.A.1-4 once prior to the period of extended operation and once within the first ten years of the period of extended operation.
46	Inspect the condition of a sample of the torus hold-down bolts and associated grout and determine appropriate actions based on the findings prior to the period of extended operation.

OI 4.2: (SER Sections: 3.0.3.2.15 - Reactor Vessel Surveillance Program, 4.2 - Reactor Vessel Neutron Embrittlement, 4.7.1 - Reflood Thermal Shock of the Reactor Vessel Internals, 4.7.2.1 BWRVIP-05, Reactor Vessel Circumferential Welds)

Due to the lack of benchmarking data in support of the plant-specific RAMA fluence calculations, the staff finds neutron fluence values unacceptable for use in the reactor vessel neutron embrittlement TLAAs.

OI 4.2 Response

OI 4.2 was clarified by the NRC in a request for additional information (RAI) transmitted in a letter dated March 26, 2007. The RAIs and responses are provided below.

RAI# 4.2

1. Fluence was calculated for the Pilgrim reactor vessel (RV) for the extended 60-year licensed operating period (54 effective full power years *(EFPY)* of facility operation), using the Radiation Analysis Modeling Application (RAMA) fluence methodology. The RAMA fluence methodology was previously approved by the NRC staff, and the results are acceptable for licensing actions provided that: (1) the RAMA application follows the guidance in Regulatory Guide 1.190 and (2) RV fluence calculations have at least one credible plant-specific surveillance capsule for benchmarking.

   The applicant provided 54 EFPY fluence values for the Pilgrim RV beltline materials in Section 4.2.1 of the License Renewal Application (LRA). These fluence values were used throughout Section 4.2 of the LRA for the RV neutron embrittlement time limited aging analyses (TLAAs). However, due to the lack of a credible plant-specific benchmark, the staff finds the 54 EFPY fluence values provided in LRA Section 4.2.1 unacceptable for use in the RV neutron embrittlement TLAAs. Therefore, the staff requests that the applicant revise Section 4.2.1 of the LRA to provide an acceptable neutron fluence evaluation or an alternative proposal for closing this TLAA topic in the LRA review.

2. Due to the lack of benchmarking data in support of the plant-specific RAMA fluence calculations, the staff cannot complete its review of the TLAAs in LRA Sections 4.2.2, 4.2.3, 4.2.4, 4.2.5, 4.2.6 and 4.7.1, as well as the aging management program (AMP) on the RV material surveillance program, using the current fluence values for the Pilgrim RV that were provided in LRA Section 4.2.1. Therefore, the staff requests that the applicant revise LRA Sections 4.2.2, 4.2.3,4.2.4, 4.2.5, 4.2.6, 4.7.1, and the AMP on the RV material surveillance program to provide an acceptable evaluation of these topics or an alternative proposal for closing these topics in the LRA review.

Response

The benchmarking validation of the RAMA fluence calculation is ongoing for the Pilgrim reactor vessel and internals. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. To ensure resolution of this issue, Commitment 47, which reads as follows, is added by this letter.

47	On or before September 15, 2007 submit to the NRC an action plan to improve benchmarking data to support approval of new P-T curves for Pilgrim.

To address this issue, an alternative analysis is provided as a means to close this TLAA topic in the LRA review. To address fluence-related TLAAs for the period of extended operation, Entergy has evaluated the affected TLAAs to determine the limiting fluence value. The evaluation included information presented in LRA sections 4.2.1, 4.2.2, 4.2.3, 4.2.4, 4.2.5, 4.2.6, 4.7.1, and the AMP on the RV material surveillance program. From this evaluation the limiting fluence was determined.

The alternative analysis to determine the limiting fluence value is included as Attachment E. This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria. This analysis also shows that there is a large margin available to this limiting fluence at the end of the period of extended operation.

The analysis determined that the limiting fluence value was set by the temperature required to perform the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400 that requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed increasing fluence and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum starting temperature for the hydrostatic test was set at 212°F with a corresponding maximum allowable 1/4T fluence of 4.12E+18 n/cm^2.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for all fluence-related TLAAs. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that all of the fluence-related TLAAs remain valid, Commitment 48, which reads as follows, is added by this letter.

48	On or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

Entergy would find it acceptable if this commitment became a license condition.

It should be noted that at the ACRS meeting on April 4, 2007, reference was made to EPRI research that investigated the irradiated behavior of stainless steel components in order to predict service life. Further review has shown that the predictions of service life related to fluence are not directly relevant in this case. The core shroud and the top guide are components that are susceptible to aging effects. However, a review of the analyses related to the core shroud found that the only time-limited aging analysis (TLAA) involves the fatigue analysis and calculation of cumulative usage factors (CUFs) for the shroud repair. The core shroud does not affect the operating P-T limit curves and there is no criterion on fluence that would further limit the operation of the core shroud structure. Similarly, the top guide does not affect the operating P-T limit curves, and there is no criterion on fluence that would further limit the operation of the top guide structure.

PNPS has re-evaluated the neutron embrittlement issues of Sections 4.2 and 4.7.1 and prepared revised LRA sections below. The Reactor Vessel Material Surveillance Program, with the changes to the fluence extrapolation, is correct as written, and no changes to Appendix B, Section B.1.26 are necessary.

<u>LRA Amendments</u>

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The regulations governing reactor vessel integrity are in 10 CFR 50. Section 50.60 requires that all light-water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50 Appendices G and H.

The PNPS current licensing basis analyses evaluating reduction of fracture toughness of the PNPS reactor vessel for 40 years are TLAA. The reactor vessel neutron embrittlement TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii) as summarized below. Fifty-four effective full-power years (EFPY) are projected for the end of the period of extended operation (60 years) assuming an average capacity factor of 90% for 60 years.

4.2.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel neutron embrittlement.

Fluence values were calculated using the RAMA fluence methodology. The RAMA fluence methodology was developed for the Electric Power Research Institute, Inc. and the boiling water reactor vessel and internals project (BWRVIP) for the purpose of calculating neutron fluence in boiling water reactor components. This methodology has been approved by the NRC (Reference 4.2-20) for application in accordance with Regulatory Guide (RG) 1.190; assuming the results are appropriately benchmarked.

The benchmarking validation of the RAMA fluence calculation is ongoing for the Pilgrim reactor vessel. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. Commitment 47 requires a plan for resolving this discrepancy to be developed and submitted for review by September 2007.

An alternative analysis to determine the limiting fluence value has been performed. This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria.

The analysis determined that the limiting fluence value was set by the temperature required to perform the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature starting for the hydrostatic test was set at 212°F The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E+18 n/cm^2. This fluence level was the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for all fluence-related TLAAs. Commitment 48 is to confirm that the limiting fluence will not be reached during the period of extended operation and consequently that all of the fluence-related TLAAs will be valid to the end of the period of extended operation.

At PNPS, the limiting beltline material for 40 years consists of 6 plates and their connecting welds, all adjacent to the active fuel zone. No nozzles are included in the limiting beltline materials for the current term of operation (Reference 4.2-2).

The beltline will be re-evaluated for 60 years. An evaluation of the RTNDT for nozzle forgings and welds is expected to show that their adjusted reference temperature at 54 EFPY will be well below the adjusted reference temperatures used in determining the P-T limits. Thus, the nozzle forgings and welds are not expected to be the limiting items for the period of extended operation.

4.2.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires that reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the Reactor Vessel Surveillance Program.

Pilgrim received License Amendment 227 dated March 29, 2007 that extended the existing P-T limit curves for Pilgrim through Cycle 18.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

4.2.3 Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb...." The initial (unirradiated) values of upper-shelf energy (CVUSE) for PNPS beltline welds were provided to the NRC in correspondence responding to Generic Letter 92-01 (References 4.2-9, 4.2-10).

Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Revision 2, provides two methods for determining Charpy upper-shelf energy (CVUSE). Position 1 applies for material that does not have surveillance data and Position 2 applies for material with surveillance data. Position 2 requires a minimum of two sets of credible material surveillance data. Since PNPS has data from only one material surveillance capsule, Position 2 does not apply. For Position 1, the percent drop in CVUSE for a stated copper content and neutron fluence is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial CVUSE to obtain the adjusted CVUSE.

The predictions for percent drop in CVUSE at 54 EFPY must be based on chemistry data, the maximum 1/4T fluence values, and unirradiated CVUSE data submitted to the NRC in the PNPS response to GL 92-01. The predicted CVUSE values for 54 EFPY will utilize Regulatory Guide 1.99 Position 1. The predictions will use Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines will be used to calculate the percent drop in CVUSE (Reference 4.2-14).

PNPS will use chemistry data from previous licensing submittals, the PNPS response to GL 92-01 (References 4.2-9, 4.2-10, 4.2-14), and the 1/4T fluence values to be determined to perform linear interpolation on the CVUSE percent drop values in RG 1.99, Revision 2, Figure 2.

The license renewal SER for BWRVIP-74 (Reference 4.2-11), Action Item #10, states that each license renewal applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials is less than that specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds given in BWRVIP-74. This action item is not applicable to PNPS if the PNPS projected CVUSE remains above the 50 ft-lb limit, even for the period of extended operation.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel Charpy upper shelf energy TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.2.4 Adjusted Reference Temperature

Irradiation by high-energy neutrons raises the value of RT_{NDT} for the reactor vessel. RT_{NDT} is the reference temperature for nil-ductility transition as defined in Section NB-2320 of the ASME Code. The initial RT_{NDT} is determined through testing of unirradiated material specimens. The shift in reference temperature, ΔRT_{NDT}, is the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. The adjusted reference temperature (ART) is defined as initial RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in RG 1.99, Revision 2. The P-T curves are developed from the ART value for the vessel materials. RG 1.99 Revision 2 defines the calculation methods for RT_{NDT} and ART.

The PNPS reactor vessel was evaluated for an assumed exposure of less than 10^{19} nvt of neutrons with energies exceeding 1 MeV (Reference 4.2-1). After approximately 4.17 EFPY, the first surveillance capsule was withdrawn from the vessel and tested. The capsule test report concludes that the shift in RT_{NDT} and upper-shelf energy over 32 EFPY will be within 10 CFR 50 guidelines.

PNPS will project values for ΔRT_{NDT} and ART at 54 EFPY using the methodology of RG 1.99. These values will be calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) contained in the PNPS response to GL 92-01 (References 4.2-3, 4.2-9, 4.2-10, 4.2-13). Initial RT_{NDT} values are from report SIR-00-082, which was submitted in 2001 as part of the PNPS P-T limit change request (Reference 4.2-5). The 1/4T fluence values discussed in Section 4.2.1 will be used. New fluence factors (FFs) will be calculated using the expression in RG 1.99, Revision 2, Equation 2, where the fluence factor is given by

$$FF = f^{(0.28-0.10*\log f)}$$

In this equation, *f* is the 1/4T fluence value. The new ΔRT_{NDT} values will be calculated by multiplying the CF and the FF for each plate and weld. Calculated margins and the initial RT_{NDT} will then be added to the calculated ΔRT_{NDT} in order to arrive at the new value of ART.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel adjusted reference temperature TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.2.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on an analysis indicating acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

PNPS received NRC approval for this relief for the remainder of the original 40-year license term. The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement (References 4.2-16, 4.2-17). The anticipated changes in metallurgical conditions expected over the extended operating period require additional analysis to extend this relief request.

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the reactor pressure vessel (RPV) shell weld failure probabilities. Three key inputs to the PFM analysis are (1) the estimated end-of-life mean neutron fluence, (2) mean chemistry values based on vessel types, and (3) the assumption of potential for beyond-design-basis events.

PNPS will compare the reactor vessel limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. The data will be from the NRC SER for PNPS Relief Request 28 (Reference 4.2-17), and from the data in Table 2.6.4 of the NRC SER for BWRVIP-05 (Reference 4.2-18). (For comparison, the EOL mean RT_{NDT} will be calculated without margin and hence will be lower than the Section 4.2.2 RT_{NDT} value.)

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when PNPS requested approval of the BWRVIP-05 technical alternative for the current license term.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel circumferential weld failure probability TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.2.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds (BWRVIP-05) are based on generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than $5x10^{-6}$ per reactor year (Reference 4.2-18). BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

PNPS received relief from the circumferential weld inspections for the remainder of the original 40-year operating term (Reference 4.2-17). The basis for this relief request was a plant-specific analysis that showed the limiting conditional failure probability for the PNPS circumferential welds at the end of the original operating term was less than the values calculated in the BWRVIP-05 SER (Reference 4.2-11). The BWRVIP-05 SER concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than $5x10^{-6}$ per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the reactor vessel axial welds will be inspected. The PNPS relief request requires additional relief request if less than 90% coverage is achieved.

PNPS will compare the reactor vessel limiting axial weld parameters to those used in the NRC analysis. The parameters used will be those from the NRC SER for BWRVIP-05 (Reference 4.2-18) from the NRC Supplemental SER for BWRVIP-05 (Reference 4.2-19).

The supplemental SER required the limiting axial weld to be compared with data found in Table 3 of the document. Originally, the supplemental SER identified PNPS as a limiting plant for the BWR fleet; however, in the discussion it is noted that the high EOL value of RT_{NDT} for PNPS calculated by the BWRVIP is due to the use of an initial RT_{NDT} of 0°F. The supplemental SER notes that the docketed value of initial RT_{NDT} (from the RVID) is -48°F, and therefore the EOL value of RT_{NDT} for PNPS is not bounding for the BWR fleet. The supplemental SER stated that the axial welds for the Clinton plant are the limiting welds for the BWR fleet and vessel failure probability determined for Clinton should bound the BWR fleet.

The limiting values will be compared to the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 supplemental SER and the 64 EFPY limits and values obtained from Table 2.6- 5 of the SER. As such, this TLAA will be projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel axial weld failure probability TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

4.7.1 Reflood Thermal Shock of the Reactor Vessel Internals

UFSAR Section 3.3.6.8 addresses reflood thermal shock of the reactor vessel internals (core shroud). This evaluation of thermal shock was considered a TLAA as it is potentially based on shroud material properties that are affected by neutron fluence.

The shroud material is Type 304 stainless steel, which is not significantly affected by irradiation.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence level was the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reflood thermal shock TLAA. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that this TLAA will be valid to the end of the period of extended operation, Commitment 48 is added.

<u>Changes to existing UFSAR Section 3.3.6.8 information presented in Section A.1.1 of the LRA (page A-3) are revised as follows:</u>

<u>3. Shroud inner surfaces at highest irradiation zone</u>

~~The most irradiated point on the inner surface of the shroud is subjected to a total integrated neutron flux of 2.7 x 10^{20} nvt (>1 MeV) by the end of station life.~~ The peak thermal shock stress is 155,700 psi, corresponding to a peak strain of 0.57 percent. The shroud material is Type 304 stainless steel, which is not significantly affected by irradiation. The material does experience a loss in reduction of area. Because reduction of area is the property which determines tolerable local strain, irradiation effects can be neglected. The peak strain resulting from thermal shock at the inside of the shroud represents no loss of integrity of the reactor vessel inner volume. ~~The service limit of Type 304 stainless steel is approached at a fluence of 8 x 1021 n/cm2 (BWRVIP-35). As the PNPS shroud will remain below that fluence level for the period of extended operation, the shroud will remain serviceable.~~

<u>UFSAR Supplement Sections are revised to read as follows:</u>

A.2.2.1.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel embrittlement. Fluence values were calculated using the RAMA fluence calculation method. The RAMA fluence method was developed for the Electric Power Research Institute, Inc. and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in boiling water reactor components. This method has been approved by the NRC (Reference A.2-9) for application in accordance with Regulatory Guide 1.190 provided the fluence calculations for the reactor are appropriately benchmarked.

The benchmarking validation of the RAMA fluence calculation is ongoing for the PNPS reactor vessel. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. An action plan to improve benchmarking data to support approval of new P-T curves will be developed and submitted for NRC review.

An alternative analysis to determine the limiting fluence value has been performed (Reference A.2-12). This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria.

The analysis determined that the limiting fluence value was set by the temperature required to perform the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature starting for the hydrostatic test was set at 212°F The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E+18 n/cm^2. This fluence level was the limiting fluence value identified.

On or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires that reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the Reactor Vessel Surveillance Program.

Pilgrim received License Amendment 227 dated March 29, 2007 that extended the existing P-T limit curves for Pilgrim through Cycle 18.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

A.2.2.1.3 Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb...." The initial (unirradiated) values of upper-shelf energy (CvUSE) for PNPS beltline welds were provided to the NRC in correspondence responding to Generic Letter 92-01.

Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Revision 2, provides two methods for determining Charpy upper-shelf energy (CvUSE). Position 1 applies for material that does not have surveillance data and Position 2 applies for material with surveillance data. Position 2 requires a minimum of two sets of credible material surveillance data. Since PNPS has data from only one material surveillance capsule, Position 2 does not apply. For Position 1, the percent drop in CvUSE for a stated copper content and neutron fluence is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial CvUSE to obtain the adjusted CvUSE.

The predictions for percent drop in CVUSE at 54 EFPY must be based on chemistry data, the maximum 1/4T fluence values, and unirradiated CvUSE data submitted to the NRC in the PNPS response to GL 92-01. The predicted CvUSE values for 54 EFPY will utilize Regulatory Guide 1.99 Position 1. The predictions will use Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines will be used to calculate the percent drop in CvUSE.

PNPS will use chemistry data from previous licensing submittals, the PNPS response to GL 92-01, and the 1/4T fluence values to be determined to perform linear interpolation on the CvUSE percent drop values in RG 1.99, Revision 2, Figure 2.

The license renewal SER for BWRVIP-74, Action Item #10, states that each license renewal applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials is less than that specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds given in BWRVIP-74. This action item is not applicable to PNPS if the PNPS projected CvUSE remains above the 50 ft-lb limit, even for the period of extended operation.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel Charpy upper shelf energy TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.4 Adjusted Reference Temperature

Irradiation by high-energy neutrons raises the value of RT_{NDT} for the reactor vessel. RT_{NDT} is the reference temperature for nil-ductility transition as defined in Section NB-2320 of the ASME Code. The initial RT_{NDT} is determined through testing of unirradiated material specimens. The shift in reference temperature, ΔRT_{NDT}, is the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. The adjusted reference temperature (ART) is defined as initial RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in RG 1.99, Revision 2. The P-T curves are developed from the ART value for the vessel materials. RG 1.99 Revision 2 defines the calculation methods for RT_{NDT} and ART.

The PNPS reactor vessel was evaluated for an assumed exposure of less than 10^{19} nvt of neutrons with energies exceeding 1 MeV. After approximately 4.17 EFPY, the first surveillance capsule was withdrawn from the vessel and tested. The capsule test report concludes that the shift in RT_{NDT} and upper-shelf energy over 32 EFPY will be within 10 CFR 50 guidelines.

PNPS will project values for ΔRT_{NDT} and ART at 54 EFPY using the methodology of RG 1.99. These values will be calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) contained in the PNPS response to GL 92-01. Initial RT_{NDT} values are from report SIR-00-082, which was submitted in 2001 as part of the PNPS P-T limit change request. The 1/4T fluence values discussed in Section 4.2.1 will be used. New fluence factors (FFs) will be calculated using the expression in RG 1.99, Revision 2, Equation 2, where the fluence factor is given by

$$FF = f^{(0.28-0.10*\log f)}.$$

In this equation, *f* is the 1/4T fluence value. The new ΔRT_{NDT} values will be calculated by multiplying the CF and the FF for each plate and weld. Calculated margins and the initial RT_{NDT} will then be added to the calculated ΔRT_{NDT} in order to arrive at the new value of ART.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel adjusted reference temperature TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on an analysis indicating acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

PNPS received NRC approval for this relief for the remainder of the original 40-year license term. The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended operating period require additional analysis to extend this relief request.

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the reactor pressure vessel (RPV) shell weld failure probabilities. Three key inputs to the PFM analysis are (1) the estimated end-of-life mean neutron fluence, (2) mean chemistry values based on vessel types, and (3) the assumption of potential for beyond-design-basis events.

PNPS will compare the reactor vessel limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. The data will be from the NRC SER for PNPS Relief Request 28, and from the data in Table 2.6.4 of the NRC SER for BWRVIP-05. (For comparison, the EOL mean RT_{NDT} will be calculated without margin and hence will be lower than the Section 4.2.2 RT_{NDT} value.)

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when PNPS requested approval of the BWRVIP-05 technical alternative for the current license term.

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel circumferential weld failure probability TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

A.2.2.1.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds (BWRVIP-05) are based on generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year. BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

PNPS received relief from the circumferential weld inspections for the remainder of the original 40-year operating term. The basis for this relief request was a plant-specific analysis that showed the limiting conditional failure probability for the PNPS circumferential welds at the end of the original operating term was less than the values calculated in the BWRVIP-05 SER. The BWRVIP-05 SER concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than 5×10^{-6} per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the reactor vessel axial welds will be inspected. The PNPS relief request requires additional relief request if less than 90% coverage is achieved.

PNPS will compare the reactor vessel limiting axial weld parameters to those used in the NRC analysis. The parameters used will be those from the NRC SER for BWRVIP-05 from the NRC Supplemental SER for BWRVIP-05.

The supplemental SER required the limiting axial weld to be compared with data found in Table 3 of the document. Originally, the supplemental SER identified PNPS as a limiting plant for the BWR fleet; however, in the discussion it is noted that the high EOL value of RT_{NDT} for PNPS calculated by the BWRVIP is due to the use of an initial RT_{NDT} of 0°F. The supplemental SER notes that the docketed value of initial RT_{NDT} (from the RVID) is -48°F, and therefore the EOL value of RT_{NDT} for PNPS is not bounding for the BWR fleet. The supplemental SER stated that the axial welds for the Clinton plant are the limiting welds for the BWR fleet and vessel failure probability determined for Clinton should bound the BWR fleet.

The limiting values will be compared to the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 supplemental SER and the 64 EFPY limits and values obtained from Table 2.6- 5 of the SER. As such, this TLAA will be projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

An analysis determined that the limiting fluence value is the fluence that corresponds to the maximum temperature limit for performing the ASME Code hydrostatic test. The temperature to perform the hydrostatic test is prescribed by ASME Section XI, Article G-2400, which requires a safety factor of 1.5 on the pressure stress intensity to prevent brittle fracture of the vessel during this test. The vessel integrity analysis assumed different fluences and calculated the corresponding hydrostatic test temperatures. As fluence increases, higher temperatures are required to perform the test to meet the ASME Code criteria. The maximum temperature for the hydrostatic test was set at 212°F. The corresponding maximum allowable fluence is a 1/4T fluence of 4.12E18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel axial weld failure probability TLAA. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy will submit to the NRC on or before June 8, 2010 calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.

<u>The following reference is added to UFSAR Supplement Section A.2.3.</u>

A.2-12 Bethay, Stephen J. (Entergy), to Document Control Desk (NRC), "License Renewal Application Amendment 16," letter 2.07.027 dated May 1, 2007, Attachment E, Structural Integrity Associates Fluence Evaluation for PNPS.

ATTACHMENT C to Letter 2.07.027
(1 page)

LRA Amendments to Delete the BWRVIP-48 and BWRVIP-49
Fatigue Assessments as TLAAs

LRA Amendments to Delete BWRVIP-48 and BWRVIP-49 Fatigue Assessments as TLAAs

Sections 4.7.2.2 and 4.7.2.3 of the draft SER discuss BWRVIP-48, Vessel ID Attachment Welds and BWRVIP-49, Instrument Penetrations, respectively. Both sections appropriately conclude that the BWRVIP analysis does not constitute a TLAA for license renewal since the CLB does not include a plant-specific 40-year CUF calculation for these components.

Section 4.7.2.3 states, "In a letter dated May 11, 2006, the applicant amended the LRA to delete the BWRVIP-49 fatigue assessment for the RPV IPNs as a TLAA for the LRA. The amendment deleted LRA Section A.2.2.6." The referenced amendment letter does not contain the stated change. Apparently the change was made to the Vermont Yankee LRA, but due to administrative oversight was not made to the PNPS LRA. The change is equally applicable to PNPS. Amendments to the LRA to delete the BWRVIP-49 fatigue assessment as a TLAA are provided.

Also, Section 4.7.2.2 should have a similar discussion of changes to the LRA necessary to delete discussion of BWRVIP-48 as a TLAA. Amendments to the LRA to delete the BWRVIP-48 fatigue assessment as a TLAA are provided.

<u>LRA Amendments to Delete the BWRVIP-48 Fatigue Assessment as a TLAA</u>

Delete Section 4.7.2.2, "BWRVIP-48, Vessel ID Attachment Welds."

Delete entry for BWRVIP-48, vessel ID attachment welds fatigue analysis from Table 4.1-1, "List of PNPS TLAA and Resolution."

Delete entry for cracking – fatigue with TLAA-metal fatigue from ID attachment welds in Table 3.1.2-1. Cracking managed by the BWR Vessel ID Attachment Welds Program remains in the table.

Delete Section A.2.2.5, "Vessel ID Attachment Welds Fatigue Analysis."

<u>LRA Amendments to Delete the BWRVIP-49 Fatigue Assessment as a TLAA</u>

Delete Section 4.7.2.3, "BWRVIP-49, Instrument Penetrations."

Delete entry for BWRVIP-49, instrument penetrations fatigue analysis from Table 4.1-1, "List of PNPS TLAA and Resolution."

Delete entry for cracking – fatigue with TLAA-metal fatigue from nozzles, reactor vessel instrumentation (N15, N16) in Table 3.1.2-1. Cracking managed by the BWR Penetrations Program remains in the table.

Delete Section A.2.2.6, "Instrument Penetrations Fatigue Analysis."